December 13, 2006

Mr. Christopher M. Crane
President and Chief Nuclear Officer
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BYRON STATION, UNIT NOS. 1 AND 2 AND BRAIDWOOD STATION, UNIT NOS. 1 AND 2 - REQUEST FOR ADDITIONAL INFORMATION RELATED TO APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY (TAC NOS. MC8966, MC8967, MC8968, AND MC8969)

Dear Mr. Crane:

By letter to the Nuclear Regulatory Commission (NRC) dated November 18, 2005, Exelon Generation Company, LLC submitted an amendment request that would revise the technical specification requirements related to steam generator tube integrity, for the Byron Station, Unit Nos. 1 and 2, and Braidwood Station, Unit Nos. 1 and 2.

The NRC staff is reviewing your submittal and has determined that additional information is required to complete the review. The specific information requested is addressed in the enclosure to this letter. During a discussion with your staff on December 5, 2006, it was agreed that you would provide a response within 60 days from the date of this letter.

The NRC staff considers that timely responses to requests for additional information help ensure sufficient time is available for staff review and contribute toward the NRC's goal of efficient and effective use of staff resources. If circumstances result in the need to revise the requested response date, please contact me at (301) 415-3733.

Sincerely,

***/RA/***

Robert F. Kuntz, Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-454, STN 50-455,
STN 50-456 and STN 50-457

Enclosure:
Request for Additional Information

cc w/encl: See next page

Mr. Christopher M. Crane
President and Chief Nuclear Officer
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

December 13, 2006

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REQUEST FOR ADDITIONAL INFORMATION

BYRON STATION, UNIT NOS. 1 AND 2,

AND BRAIDWOOD STATION, UNIT NOS. 1 AND 2

DOCKET NOS. STN 50-454, STN 50-455,

STN 50-456 AND STN 50-457

In reviewing the Exelon Generation Company's (EGC) submittal dated November 18, 2005, as supplemented by letters dated August 18 and September 28, 2006, related to a proposed amendment to revise the technical specification requirements related to steam generator tube integrity, for the Byron Station, Unit Nos. 1 and 2 (Byron) and Braidwood Station, Unit Nos. 1 and 2 (Braidwood), the NRC staff has determined that the following information is needed in order to complete its review:

References:

1. EGC letter RS-05-129, "Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity," dated November 18, 2005.

2. EGC letter RS-06-109, "Response to Request for Additional Information Regarding Application for Steam Generator Tube Integrity Technical Specification," dated August 18, 2006.

Requested Information:

1. Regarding EGC's response to the request for information (RAI) question 8 (for Byron/Braidwood, question 5 for Seabrook) in Reference 2, Attachment 6, pages 8 to 16, provide a plot of crack-opening angle for circumferential cracks located 4 inches from the bottom of the tubesheet as a function of crack length for normal operating and main steam line break conditions. Also, provide a plot of leak rate as a function of the same parameters, neglecting the effect of crevice resistance.

2. Regarding EGC's response to RAI question 8 (for Byron/Braidwood, question 5 for Seabrook) in Reference 2, Attachment 6, pages 8 to 16, provide revised versions of Figures 3 and 4 to include the leak rate ratios for cracks in the range of 0.1 to 0.5 inches in length. It would seem from Figures 3 and 4, that if crack resistance dominates crevice resistance, then leakage ratios may exceed 2 for through wall crack lengths less than 0.5 inches for tubes near the periphery of the bundle, particularly for circumfrential cracks. Also, provide similar figures for the near radius and mid-radius locations.

3. The discussion accompanying Figures 3 and 4 states that cracks less than 0.5 inches in length are not expected to cause any "relative significant leakage." Please explain basis for concluding the leakage contribution from the population of circumferential cracks of

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through-wall length less than 0.5 inches is small, relative to the leakage contribution from the population of through-wall cracks greater than 0.5 inches in length such that the leakage ratio between normal operating and accident conditions is dominated by the leakage ratio (which is less than 2) exhibited by the population of cracks larger than 0.5 inches. This explanation should consider any relevant operating experience regarding the probability density function of 100 percent through wall crack lengths and, in addition, the plots provided in response to question 2 above.

4. Regarding a statement in the discussion underneath Figure 4 that reads, "The results from the crack-only analyses show that in the absence of the dent the resistance to flow is increased and each crack type produces a lower leak rate ratio," please clarify what is meant by the "dent," and its impact on this statement. Additionally, please qualify what the increase in resistance to flow and lower leak ratios are relative to.

5. Reference 2, Attachment 6, page 12, Analysis of Circumferential Cracking, states that the circumferential crack model was developed in WCAP-15932-P, Revision 1, "Improved Justification of Partial Length RPC inspection of Tube Joints of Model F Steam Generators of Ameren-UE Callaway Plant," dated May 2003. WCAP-15932P, Appendix C states that the main loadings on a circumferential crack below the H* distance are the pressure loads acting on the crack face. It is also stated in the WCAP that the internal pressure end cap load is not transmitted below about 1/3 the H* distance. Assuming that H* is determined correctly, the NRC staff agrees that this statement is true for normal operating pressure provided the tube is severed immediately below the 1/3 H* distance. Similarly, the 3 delta P end cap load does not extend below the full H* distance assuming the tube is severed immediately below the H* distance. If the tube is not severed, then much of the end cap load will be transmitted below the H* distance. Taking an extreme example, the calculated H* distance is based in part on pull out tests (on specimens that were basically severed at the bottom) where the pull out criterion was an axial displacement of 0.25 inches at the bottom of the specimen. If the tube is intact below the H* distance, then the tube must be able to stretch by 0.25 inches between the weld and the H* location which means there must be considerable force transmitted below the H* distance. For smaller end cap loads where no slippage takes place, a severed tube end would be expected to displace upward due to the accumulated strains in the tube to tubesheet joint above the severed location. If the tube is not completely severed, the tube below the crack would be expected to resist this displacement and thus resist some of the pullout load. The tube to tubesheet joint (where the tube is not severed inside the tubesheet) is a redundant structure. How much of the end cap load that gets transmitted below the crack location (assumed to be 17 inches down from the top of the tubesheet) depends on the stiffness of the friction joint above the crack relative to the stiffness of the tube below the crack. It is not clear from the NRC staff's review of the model that this effect has been evaluated. Thus, it is not clear to the NRC staff that the axial load acting on the circumferentially-cracked cross section is limited solely to the pressure acting on the crack faces and that no portion of the internal pressure end cap load is acting on the cross section. Please address this concern, including how the stiffness of the tube to tubesheet friction joint above the crack relative to the stiffness of the tube below the crack have been specifically accounted for. Has a detailed analysis (e.g., finite element analysis) been performed to determine how much of the full internal pressure end cap

load is actually transmitted to the cracked cross section under normal operating and accident conditions? If so, describe the analysis and the results.

6. The Reference 1 application included the following provision in TS 5.5.9c: "For Unit 2 only, degradation found in the portion of the tube from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet shall be plugged or repaired upon detection." In the example accompanying the NRC staff's draft RAI No. 4, the NRC staff inadvertently left this sentence out. It wasn't the NRC staff's intent to suggest this sentence should be deleted. Describe your plan for re-including this sentence as part of TS 5.5.9c. Also, as a point of consistency and clarification, the word "degradation" in the above sentence and in TS5.5.9c.4.i should be replaced by the word "flaws" consistent with the rest of the technical specifications. Please describe your plan for making this change as well.

7. Did any of the hydraulic expansions in the Model D5 SGs experience a stress relief during fabrication, directly or indirectly (e.g., as a result of stress relieving the shell to tubesheet welds)? If so, how was this reflected in the pullout and leakages tests in support of the tubesheet amendment requests?

8. The tubesheet bow analysis described in Westinghouse report, LTR-CDME-05-32-P, Rev 2, submitted as Attachment 7 to Reference 1, takes credit for resistance against bow provided by the divider plate. Cracks in the welds connecting the tubesheet and divider plate have been found by inspection at certain foreign steam generators. Please discuss how such cracks, if present at the Byron/Braidwood units, could affect the conservatism of the proposed 17-inch tubesheet inspection distance requirement for ensuring structural and leakage integrity.